ATTACHMENT B

MARKED UP PAGES FOR
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS FOR
FACILITY OPERATING LICENSES
NPF-37, NPF-66, NPF-72, AND NPF-77

Revised Pages for:

BYRON NPF-37 AND NPF-66	BRAIDWOOD NPF-72 AND NPF-77
3/4 4 -13*	3/4 4 -13*
3/4 4 -14	3/4 4 -14
------------	3/4 4 -14a*
3/4 4 -15*	3/4 4 -15*
3/4 4 -16	3/4 4 -16
3/4 4 -17*	3/4 4 -17*
3/4 4 -17a	3/4 4 -17a
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B 3/4 4 - 3	B 3/4 4 - 3
B 3/4 4 - 3a	B 3/4 4 - 3a

Pages indicated with an asterisk(*) are not being revised and are only included for convenience.

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3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable steam generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube* Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. When applying the expectations of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by the sleeve are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

*When referring to a steam generator tube, the sleeve shall be considered a part of the tube if the tube has been repaired per Specification 4.4.5.4.a.10.

SURVEILLANCE REQUIREMENTS (Continued)

1) All tubes that previously had detectable tube wall penetrations greater than 20% that have not been plugged or sleeved in the affected area, and all tubes that previously had detectable sleeve wall penetrations that have not been plugged,

2) Tubes in those areas where experience has indicated potential problems,

3) At least 3% of the total number of sleeved tubes in all four steam generators or all of the sleeved tubes in the generator chosen for the inspection program, whichever is less. These inspections will include both the tube and the sleeve, and

4) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

5) For Unit 1, tubes left in service as a result of application of the tube support plate plugging criteria shall be inspected by bobbin coil probe during all future outages.

INSERT A

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

d. For Unit 1, Cycle 7 implementation of the tube support plate interim plugging criteria limit requires a 100% bobbin coil probe inspection for all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with outer diameter stress corrosion cracking (ODSCC) indications. The determination of the tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of tubes inspected over their full length.

e. A random sample of at least 20 percent of the total number of sleeves shall be inspected for axial and circumferential indications at the end of each cycle. In the event that an imperfection of 40 percent or greater depth is detected, an additional 20 percent of the unsampled sleeves shall be inspected, and if an imperfection of 40 percent or greater depth is detected in the second sample, all remaining sleeves shall be inspected. These inservice inspections will include the entire sleeve and the tube at the heat treated area. The inservice inspection for the sleeves is required until the corrosion resistance for the laser welded or kinetically welded joints in tubes that bound the material parameters of the tubes installed in the steam generators has been demonstrated acceptable. If conformance with the acceptable criteria

INSERT A

BYRON INSERT

6) For Unit 1, tubes which remain in service due to the application of the F* criteria will be inspected, in the tubesheet region, during all future outages.

BRAIDWOOD INSERT

5) For Unit 1, tubes which remain in service due to the application of the F* criteria will be inspected, in the tubesheet region, during all future outages.

SURVEILLANCE REQUIREMENTS (Continued)

of Specification 4.4.5.4 for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes or sleeves must exhibit significant (greater than 10% of wall thickness) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1) Reactor-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2c., or

d) Certain intersections as identified in WCAP-14046, Section 4.7, will be excluded from application of the voltage-based repair criteria as it is determined that these intersections may collapse or deform following a postulated LOCA+SSE event.

e) If, as a result of leakage due to a mechanism other than ODSCC at the tube support plate intersection, or some other cause, an unscheduled mid-cycle inspection is performed, the following repair criteria apply instead of 4.4.5.4.11)c). If bobbin voltage is within expected limits, the indication can remain in service. The expected bobbin voltage limits are determined from the following equation:

$$V < \frac{\frac{\Delta t}{CL}(V_{SL} - V_{BOC}) + V_{BOC}}{1 + (0.2)\left(\frac{\Delta t}{CL}\right)}$$

where:

V	=	measured voltage
V_{BOC}	=	voltage at BOC
Δt	=	time period of operation to unscheduled outage
CL	=	cycle length (full operating cycle length where operating cycle is the time between two scheduled steam generator inspections)
V_{SL}	=	4.5 volts

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b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair in the affected area all tubes exceeding the plugging or repair limit) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged or repaired.

INSERT C

12) <u>F* Distance</u> is the distance into the tubesheet from the secondary face of the tubesheet or the top of the last hardroll, whichever is further into the tubesheet, that has been determined to be 1.7 inches.

13) <u>F* Tube</u> is a Unit 1 SG tube with degradation below the F* distance and has no indications of degradation (i.e., no indication of cracking) within the F* distance. Defects contained in an F* tube are not dependant on flaw geometry.

SURVEILLANCE REQUIREMENTS (Continued)

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For Unit 1 Cycle 7, implementation of the voltage-based repair criteria to tube support plate intersections, reports to the Staff shall be made as follows:

1) Notify the Staff prior to returning the steam generators to service should any of the following conditions arise:

a) If estimated leakage based on the actual measured end-of-cycle voltage distribution would have exceeded the leak limit (for postulated main steam line break utilizing licensing basis assumptions) during the previous operation cycle.

b) If circumferential crack-like indications are detected at the tube support plate intersections.

c) If indications are identified that extend beyond the confines of the tube support plate.

d) If the calculated conditional burst probability exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

2) The final results of the inspection and the tube integrity evaluation shall be reported to the Staff pursuant to Specification 6.9.2 within 90 days following restart.

INSERT D

INSERT D

e. The results of inspections of F* Tubes shall be reported to the Commission prior to the resumption of plant operation. The report shall include:

1) Identification of F* Tubes, and

2) Location and size of the degradation.

REACTOR COOLANT SYSTEM

BASES

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown, mainsteam lines, or the steam jet air ejecters. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired by sleeving. The technical bases for sleeving are described in the current Westinghouse or Babcock & Wilcox Nuclear Technologies Technical Reports.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the plugging or repair limit of 40% of the tube nominal wall thickness, excluding defects that meet the criteria for F* tubes. If a sleeved tube is found to contain a through wall penetration in the sleeve of equal to or greater than 40% of the nominal wall thickness, the tube must be plugged. The 40% plugging limit for the sleeve is derived from Reg. Guide 1.121 analysis and utilizes a 20% allowance for eddy current uncertainty and additional degradation growth. Inservice inspection of sleeves is required to ensure RCS integrity. Sleeve inspection techniques are described in the current Westinghouse or Babcok & Wilcox Nuclear Technologies Technical Reports. Steam Generator tube and sleeve inspections have demonstrated the capability to reliably detect degradation that has penetrated 20% of the pressure retaining portions of the tube or sleeve wall thickness. Commonwealth Edison will validate the adequacy of any system that is used for periodic inservice inspection of the sleeves and, as deemed appropriate, will upgrade testing methods as better methods are developed and validated for commercial use.

3/4.4.5 STEAM GENERATORS (Continued)

For Unit 1 Cycle 7, tubes experiencing outer diameter stress corrosion cracking within the thickness of the tube support plates will be dispositioned in accordance with Specification 4.4.5.4.a.11. The operating period may be adjusted to less than the full operating cycle to meet the maximum site allowable primary-to-secondary leakage limit for End of Cycle Main Steam Line Break conditions. ~~The leakage limit, 12.8 gpm, includes the accident leakage from a faulted steam generator and the operational leakage of the three remaining intact steam generators equal to the Specification 4.4.6.2.c leakage limit.~~

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Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.9.2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

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2) A seismic occurrence greater than the Operating Basis Earthquake, or

3) A Condition IV loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

4) A Condition IV main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve;

3) Degraded Tube means a tube or sleeve containing unrepaired imperfections greater than or equal to 20% of the nominal tube or sleeve wall thickness caused by degradation;

4) % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing an unrepaired defect is defective;

6) Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair limit imperfection depth ~~is equal to~~ 40% of the nominal wall thickness. INSERT B

For Unit 1 Cycle 7, this definition does not apply to tube support plate intersections for which the voltage-based plugging criteria are being applied. Refer to 4.4.5.4.a.11 for the repair limit applicable to these intersections;

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube that has been repaired by sleeving, the tube inspection shall include the sleeved portion of the tube, and

INSERT B

For Unit 1, this definition does not apply to defects in the tubesheet that meet the criteria for an F* tube;

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

10) Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs will be performed by the following processes:

a) Laser welded sleeving as described in a Westinghouse Technical Report currently approved by the NRC, subject to the limitations and restrictions as noted by the NRC staff, or

b) Kinetic welded sleeving as described in a Babcock & Wilcox Nuclear Technologies Technical Report currently approved by the NRC, subject to the limitations and restrictions as noted by the NRC staff.

Tube repair includes the removal of plugs that were previously installed as a corrective or preventative measure. A tube inspection per 4.4.5.4.a.8 is required prior to returning previously plugged tubes to service.

11) For Unit 1 Cycle 7, the Tube Support Plate Interim Plugging Criteria Limit is used for the disposition of a steam generator tube for continued service that is experiencing outer diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the repair limit is based on maintaining steam generator tube serviceability as described below:

a) Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage less than or equal to 1.0 volt will be allowed to remain in service.

b) Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage greater than 1.0 volt will be repaired or plugged except as noted in 4.4.5.4.a.11)c) below.

c) Indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 1.0 volt but less than or equal to 2.7 volts may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of outside diameter stress corrosion cracking degradation with bobbin voltage greater than 2.7 volts will be plugged or repaired.

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable steam generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube* Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. When applying the expectations of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by the sleeve are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

*When referring to a steam generator tube, the sleeve shall be considered a part of the tube if the tube has been repaired per Specification 4.4.5.4.a.10.

SURVEILLANCE REQUIREMENTS (Continued)

1) All tubes that previously had detectable tube wall penetrations greater than 20% that have not been plugged or sleeved in the affected area, and all tubes that previously had detectable sleeve wall penetrations that have not been plugged,

2) Tubes in those areas where experience has indicated potential problems,

3) At least 3% of the total number of sleeved tubes in all four steam generators or all of the sleeved tubes in the generator chosen for the inspection program, whichever is less. These inspections will include both the tube and the sleeve, and

4) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

INSERT A

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

d. For Unit 1 Cycle 5, implementation of the tube support plate interim plugging criteria limit requires a 100% bobbin coil probe inspection for all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with outer diameter stress corrosion cracking (ODSCC) indications. An inspection using a rotating pancake coil (RPC) probe is required in order to show OPERABILITY of tubes with flaw-like bobbin coil signal amplitudes greater than 1.0 volt but less than or equal to 2.7 volts. For tubes that will be administratively plugged or repaired, no RPC inspection is required. The RPC results are to be evaluated to establish that the principal indications can be characterized as ODSCC.

e. A random sample of at least 20 percent of the total number of sleeves shall be inspected for axial and circumferential indications at the end of each cycle. In the event that an imperfection of 40 percent or greater depth is detected, an additional 20 percent of the unsampled sleeves shall be inspected, and if an imperfection of 40 percent or greater depth is detected in the second sample, all remaining sleeves shall be inspected. These inservice inspections will include the entire sleeve and the tube at the heat treated area. The inservice inspection for the sleeves is required until the corrosion resistance for the laser welded or kinetically welded joints in tubes that bound the material parameters of the tubes installed in the steam generators has been demonstrated

INSERT A

BYRON INSERT

6) For Unit 1, tubes which remain in service due to the application of the F* criteria will be inspected, in the tubesheet region, during all future outages.

BRAIDWOOD INSERT

5) For Unit 1, tubes which remain in service due to the application of the F* criteria will be inspected, in the tubesheet region, during all future outages.

SURVEILLANCE REQUIREMENTS (Continued)

acceptable. If conformance with the acceptable criteria of Specification 4.4.5.4 for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes or sleeves must exhibit significant (greater than 10% of wall thickness) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1) Reactor-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2c., or

   2) A seismic occurrence greater than the Operating Basis Earthquake, or

   3) A Condition IV loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4) A Condition IV main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve;

3) Degraded Tube means a tube or sleeve containing unrepaired imperfections greater than or equal to 20% of the nominal tube or sleeve wall thickness caused by degradation;

4) % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing an unrepaired defect is defective;

6) Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair limit imperfection depth is equal to 40% of the nominal wall thickness. Insert B ~~For Unit 1 Cycle 5, this definition does not apply to the region of the tube subject to the tube support plate interim plugging criteria limit, i.e., the tube support plate intersections. Specification 4.4.5.4.a.11 describes the repair limit for use within the tube support plate intersection of the tube;~~

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube that has been repaired by sleeving, the tube inspection shall include the sleeved portion of the tube, and

INSERT B

For Unit 1, this definition does not apply to defects in the tubesheet that meet the criteria for an F* tube;

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

10) Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs will be performed by the following processes:

a) Laser welded sleeving as described by Westinghouse report WCAP-13698, Rev. 1, or

b) Kinetic welded sleeving as described by Babcock & Wilcox Topical Report BAW-2045PA, Rev. 1.

Tube repair includes the removal of plugs that were previously installed as a corrective or preventative measure. A tube inspection per 4.4.5.4.a.8 is required prior to returning previously plugged tubes to service.

11) Tube Support Plate Interim Plugging Criteria Limit for Unit 1 Cycle 5 is used for the disposition of a steam generator tube for continued service that is experiencing ODSCC confined within the thickness of the tube support plates. For application of the tube support plate interim plugging criteria limit, the tube's disposition for continued service will be based upon standard bobbin coil probe signal amplitude of flaw-like indications. The plant specific guidelines used for all inspections shall be consistent with the eddy current guidelines in Appendix A of WCAP-13854 as appropriate to accommodate the additional information needed to evaluate tube support plate signals with respect to the voltage parameters as specified in Specification 4.4.5.2. Pending incorporation of the voltage verification requirements in ASME standard verifications, an ASME standard calibrated against the laboratory standard will be utilized in Unit 1 steam generator inspections for consistent voltage normalization.

1. A tube can remain in service with a flaw-like bobbin coil signal amplitude of less than or equal to 1.0 volt, regardless of the depth of the tube wall penetration, provided Item 3 below is satisfied.

2. A tube can remain in service with a flaw-like bobbin coil signal amplitude greater than 1.0 volt but less than or equal to 2.7 volts provided an RPC inspection does not detect degradation and provided Item 3 below is satisfied.

3. The projected end of cycle distribution of crack indications is verified to result in total primary to secondary leakage less than 9.1 gpm (includes operational and accident leakage). The basis for determining expected leak rates from the projected crack distribution is provided in WCAP-14046, "Braidwood Unit 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria" dated May 1994.

4. A tube with a flaw-like bobbin coil signal amplitude of greater than 2.7 volts shall be plugged or repaired.

Certain tubes identified in WCAP-14046, "Braidwood Unit 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria," dated May 1994, shall be excluded from application of the tube support plate interim plugging criteria limit. It has been determined that these tubes may collapse or deform following a postulated LOCA + SSE.

Insert C

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair in the affected area all tubes exceeding the plugging or repair limit) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

INSERT C

12) F* Distance is the distance into the tubesheet from the secondary face of the tubesheet or the top of the last hardroll, whichever is further into the tubesheet, that has been determined to be 1.7 inches.

13) F* Tube is a Unit 1 SG tube with degradation below the F* distance and has no indications of degradation (i.e., no indication of cracking) within the F* distance. Defects contained in an F* tube are not dependant on flaw geometry.

SURVEILLANCE REQUIREMENTS (Continued)

d. For Unit 1 Cycle 5, the results of inspection for all tubes in which the tube support plate interim plugging criteria limit has been applied shall be reported to the Commission pursuant to Specification 6.9.2 following completion of the steam generator tube inservice inspection and prior to Cycle 5 operation. The report shall include:

1. Listing of the applicable tubes,

2. Location (applicable intersections per tube) and extent of degradation (voltage), and

3. Projected Steam Line Break (MSLB) Leakage.

Insert D

INSERT D

e. The results of inspections of F* Tubes shall be reported to the Commission prior to the resumption of plant operation. The report shall include:

1) Identification of F* Tubes, and

2) Location and size of the degradation.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown, mainsteam lines, or the steam jet air ejectors. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired by sleeving. The technical bases for sleeving are described in the current Westinghouse or Babcock & Wilcox Nuclear Technologies Technical Reports. 4

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the plugging or repair limit of 40% of the tube nominal wall thickness. If a sleeved tube is found to contain a through wall penetration in the sleeve of equal to or greater than 40% of the nominal wall thickness, the tube must be plugged. The 40% plugging limit for the sleeve is derived from Reg. Guide 1.121 analysis and utilizes a 20% allowance for eddy current uncertainty and additional degradation growth. Inservice inspection of sleeves is required to ensure RCS integrity. Sleeve inspection techniques are described in the current Westinghouse or Babcock & Wilcox Nuclear Technologies Technical Reports. Steam Generator tube and sleeve inspections have demonstrated the capability to reliably detect degradation that has penetrated 20% of the pressure retaining portions of the tube or sleeve wall thickness. Commonwealth Edison will validate the adequacy of any system that is used for periodic inservice inspection of the sleeves and, as deemed appropriate, will upgrade testing methods as better methods are developed and validated for commercial use.

, excluding defects that meet the criteria for F* tubes

3/4.4.5 STEAM GENERATORS (continued)

For Unit 1 Cycle 5, tubes experiencing outer diameter stress corrosion cracking within the thickness of the tube support plates will be dispositioned in accordance with Specification 4.4.5.4.a.11.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.9.2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

INSERT E

INSERT F

INSERT E

For Unit 1, plugging or repair is not required for tubes with degradation within the tubesheet area which fall under the alternate tube plugging criteria defined as F*. The F* Criteria is based on "Babcock & Wilcox Nuclear Technologies (BWNT) Topical Report BAW-10196 P."

F* tubes meet the structural integrity requirements with appropriate margins for safety as specified in Regulatory Guide 1.121 and the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB and Division I Appendices, for normal operating and faulted conditions.

INSERT F

BYRON INSERT

The leakage limit, 12.8 gpm, includes the accident leakage from IPC in addition to the accident leakage from F* on the faulted steam generator and the operational leakage limit of Specification 3.4.6.2.c. The operational leakage limit of Specification 3.4.6.2.c in each of the three remaining intact steam generators shall include the operational leakage from F*.

BRAIDWOOD INSERT

The leakage limit, 9.1 gpm, includes the accident leakage from IPC in addition to the accident leakage from F* on the faulted steam generator and the operational leakage limit of Specification 3.4.6.2.c. The operational leakage limit of Specification 3.4.6.2.c in each of the three remaining intact steam generators shall include the operational leakage from F*.

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS
FOR PROPOSED CHANGES TO
APPENDIX A TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSE
NPF-37, NPF-66, NPF-72, AND NPF-77

A. INTRODUCTION

Commonwealth Edison (ComEd) has evaluated the proposed amendment and determined that it involves no significant hazards consideration for the F* criteria application at Byron and Braidwood. The F* criteria is applicable to the Unit 1 Model D4 steam generators (SG) for Byron and Braidwood.

An alternate repair criteria (ARC) designated as the F*criteria, is being requested by ComEd to allow tubes with otherwise pluggable indications, to remain in service as long as the indications are below the designated minimum distance of the F* criteria. The F* criteria defines a length of 1.7 inches of undegraded expanded tube within the tubesheet as the minimum distance acceptable for implementing this ARC. Below the F* length, a circumferential tube defect can exist and the tube can remain in service. The proposed amendment will change the plugging limit definition and would exclude plugging steam generator tubes with indications that satisfy the F* criteria. The F* criteria maintains the structural integrity of the degraded tube as the primary pressure boundary and allows the tube to remain in service for heat transfer and core cooling.

Description of the F* Criteria and Qualification of F*:

Babcock & Wilcox Nuclear Technologies (BWNT) F* Qualification Report documents the design requirements, design verification testing results, analysis results, eddy current verification testing, and tubesheet corrosion evaluation performed to justify the F* criteria. The F* criteria defines the minimum length of undegraded tubing (i.e., no indication of cracking) within the tubesheet, below which, a tube with an otherwise pluggable defect can remain inservice. The tube defect is assumed to be the worst case condition of a 360 degree circumferential tube sever. With the worst case tube defect at the F* length, the joint exhibited sufficient strength to carry normal and faulted loads with a margin of safety consistent with Regulatory Guide 1.121 requirements and limited primary to secondary leakage within Technical Specification limits for both normal operating and faulted conditions. The F* length was determined to be 1.7 inches, which includes an eddy current uncertainty measurement. The F* length is measured from the top of the tubesheet or the top of the last hardroll, whichever is further into the tubesheet, to the top of the defect.

The F* qualification consisted of establishing tube loading, mechanical testing (tube tensile and pull out tests), leak testing, determination of eddy current uncertainty, and evaluating the effects of boric acid on the tubesheet. The tube mock-up blocks for mechanical and leak testing simulated the wide range of actual tube-to-tubesheet interface conditions in the D-4 steam generators. Various installation parameters that were considered include varying tubesheet bore sizes, varying tubesheet bore surface finish conditions, high and low yield strength tubing, and effects of thermal cycling.

Tube loading was determined for normal operating, faulted and locked tube conditions. Radial and axial stresses were determined considering the effects due to tube springback, thermal growth, tubesheet bow, primary internal pressure, and tube end forces. These stresses were determined by analytical methods and by physical tube mock-up testing. Margins of safety in accordance with Regulatory Guide 1.121 requirements were factored into the tube loading. The tube loading results were used to determine the mock-up testing parameters and used in the F* distance calculation. The F* distance calculation was derived from tube pullout equations that involve tube loading and frictional considerations for the faulted and tested conditions. The final F* length calculation was reduced to an equation that ratios the tested joint strength and stresses to the actual steam generator conditions for a given tested engagement length.

Mechanical testing included four tube loading tests to determine the strength of the rolled tube-to-tubesheet joint for various engagement lengths. The tests are listed below.

- Tensile tests that bounded the normal operating and faulted loads including the Regulatory Guide 1.121 safety margins .

- Locked Tube loading tests.

- Pressure Cycling and Axial Load Cycling tests.

- Ultimate Load Tests where the tube joints are loaded until failure.

The results of the engagement length tests were used in the final F* calculation. The joint strength value used in the final F* calculation was conservatively the lower 95% tolerance limit of the ultimate load test results.

Leak rate testing monitored the primary to secondary leakage of tubes with various engagement lengths at normal operating and faulted differential pressures. Leakage was also monitored at faulted differential pressures following pressure cycles at nominal operating pressure. The average leakage from all three tests, if applied to all tube ends, for all engagements lengths were well below the Technical Specification and administrative limits. The leakage is still within the Technical Specification limits for all tube ends if an upper 95% confidence interval were applied to the leakage results.

Eddy current uncertainty was determined through testing with equipment used during routine steam generator tube inspections. Bobbin coil and rotating pancake coil probe results were compared.

The final F* length was determined by inputting the tested joint strength and length, with the 95% tolerance limit, into the F* equation. The final F* length is 1.7 inches including eddy current uncertainties.

An evaluation was performed that concluded the effect of borated primary water leaking through the tube defect on the carbon steel tubesheet is insignificant. Due to the tight crack characteristics of Primary Water Stress Corrosion Cracking, the amount of borated solution in contact with the tubesheet is very low and the amount of oxygen in the primary water is also very low, thus minimizing the corrosive effects of boric acid. The operating temperature at the tubesheet also minimizes the corrosive effects of boric acid.

B. 10 CFR 50.92 ANALYSIS

Evaluation of F* for Significant Hazards:

According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The proposed F* criteria requires an amendment to modify the Byron and Braidwood Station Technical Specifications for inservice inspection of the steam generators. The incorporation of this criteria into the Technical Specification changes none of the original plant design conditions or performance characteristics.

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The supporting qualification report for the subject criteria demonstrates that the presence of the tubesheet will enhance the tube integrity in the region of the tube-to-tubesheet roll expansions by precluding tube deformation beyond its initial expanded outside diameter. The resistance to a tube rupture is strengthened by the presence of the tubesheet in that region. The results of hardrolling of the tube into the tubesheet provides a mechanical leak limiting seal between the tube and the tubesheet. A tube rupture cannot occur because the contact between the tube and the tubesheet does not permit sufficient movement of tube material.

   The type of degradation for which the F* criteria has been developed (cracking with a circumferential orientation) can theoretically lead to a postulated tube rupture event provided that the postulated through-wall circumferential crack exists near the top of the tubesheet. An evaluation including analysis and testing has been done to determine the resistive strength of the expanded tubes within the tubesheet. This evaluation provides the basis for the acceptance criteria for tube degradation subject to the F* criteria.

   The F* length of roll expansion is sufficient to preclude tube pullout from tube degradation located below the F* distance, regardless of the extent of the tube degradation. The Technical Specification leakage rate requirements and accident analysis assumptions remain unchanged in the unlikely event that significant leakage from this region does occur. For consistency with current offsite dose limits, the site allowable leakage limit during a MSLB has been conservatively calculated to be 12.8 gpm for Byron and 9.1 gpm for Braidwood, which includes the accident leakage from IPC in addition to the accident leakage from F* on the faulted steam generator and the operational leakage limit. The operational leakage limit of Specification 3.4.6.2.c in each of the three remaining intact steam generators shall include the operational leakage from F*. As a requirement for operation following application of IPC, the projected distribution of crack indications over the operating period must be verified to result in primary to secondary accident leakage less than the site allowable leakage limit. Thus, the consequences of a MSLB remain unchanged.

The tube rupture and pullout is fully bounded by the existing steam generator tube rupture analysis included in the UFSAR. The leakage testing of the roll expanded tubes indicates that for tube expansion lengths approximately equal to the F* distance, any postulated primary to secondary leakage from F* tubes would be insignificant. The proposed alternate repair criteria does not adversely impact any other previously evaluated design basis accident.

The leakage from an F* tube would be limited by the tube-to-tubesheet interface since this leak would occur below the secondary face of the tubesheet. Qualification testing and previous experience indicate that normal and faulted leakage is well below Technical Specification and administrative limits creating no increase in the consequences associated with tube rupture type leakages. The UFSAR analyzed accident scenarios are still bounding since the normal and faulted leak rates are well within the normal operating limit of 150 gallons per day. This conclusion is consistent with previous F* programs approved and used at other operating plants.

All of the design and operating characteristics of the steam generator and connected systems are preserved since the F* criteria utilizes the "as rolled" tube configuration that exists as part of the original steam generator design. The F* joint has been analyzed and tested for design, operating, and faulted condition loadings in accordance with Regulatory Guide 1.121 safety factors. The potential for a tube rupture is not increased from the original submittal as demonstrated in the qualification analyses and testing completed in the BWNT report.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different type of accident from any accident previously evaluated.

Implementation of the proposed F* criteria does not introduce any changes to the plant design basis. Use of the criteria does not provide a mechanism to initiate an accident outside of the region of the expanded portion of the tube. In the unlikely event the failed tube severed completely at a point below the F* region, the remaining F* joint would retain engagement in the tubesheet due to its length of expanded contact within the tubesheet bore. This engagement length would prevent any interaction of the severed tube with neighboring tubes. Any hypothetical accident as a result of any tube degradation in the expanded region of the tube would be bounded by the existing tube rupture accident analysis. Tube bundle structural integrity will be maintained. Tube bundle leak tightness will be maintained such that any postulated accident leakage from F* tubes will be negligible with regard to offsite doses.

Therefore, there is not a potential for creating the possibility of a new or different type of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety.

The use of the F* criteria has been demonstrated to maintain the integrity of the tube bundle commensurate with the requirements of Regulatory Guide 1.121 and the primary to secondary pressure boundary under normal and postulated accident conditions. Acceptable tube degradation for the F* criteria is any degradation indication in the tubesheet region, more than the F* distance from the secondary face of the tubesheet or the top of the last hardroll contact point whichever is further into the tubesheet. The safety factors used in the verification of the strength of the degraded tube are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code and Regulatory Guide 1.121 used in steam generator design. The F* distance has been verified by various testing to be greater than the length of the roll expanded tube-to-tubesheet interface required to preclude both tube pullout and significant leakage during normal and postulated accident conditions. The protective boundaries of the steam generator continue to be maintained with the use of the F* criteria. A tube with the indication of degradation previously requiring removal from service can be kept in service through the F* criteria. Since the joint is constrained within the tubesheet bore, there is no additional risk associated with the previously analyzed tube rupture event. The leak testing acceptance criteria are based on the primary to secondary leakage limit in the Technical Specifications and the leakage assumptions used in the UFSAR accident analyses.

Implementation of the alternate repair criteria will decrease the number of tubes which must be taken out of service with tube plugs or repaired by sleeves. Both plugs and sleeves reduce the RCS flow margin; thus, implementation of the F* criteria will maintain the margin of flow that would otherwise be reduced in the event of increased plugging or sleeving.

The use of the F* criteria described herein, to maintain tubes in service, does not represent an unanalyzed safety concern. Furthermore, its use does not create the possibility of a new or different type of accident from any accident previously evaluated nor does it reduce the margin of safety.

Therefore, based on the above evaluation, Commonwealth Edison has concluded that these changes do not involve any significant hazards considerations.